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June 20, 2000

SVP-00-109

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Quad Cities Nuclear Power Station, Unit 2
Facility Operating License No. DPR-30
NRC Docket No 50-265

Subject: Reactor Scram and Reactor Water Cleanup System Isolation While
Returning Turbine Control Valve to Service

Enclosed is Licensee Event Report (LER) 265/00-007, Revision 00, for Quad Cities Nuclear Power Station.

This report is submitted in accordance with the requirements of the Code of Federal Regulations, Title 10, Part 50.73(a)(2)(iv). The licensee shall report any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature.

We are committing to the following actions:

Appropriate changes to the processes and procedures associated with risk mitigation, pre-job briefs, and infrequently performed evolutions will be implemented to ensure that appropriate checks and balances are incorporated in the processes.

Documented senior manager nuclear fundamentals, with improvements based on a gap analysis, will be implemented. An action plan for improvement will be developed and implemented.

A review of recent Technical Specification events and causal factors at Quad Cities Nuclear Power Station will be developed and implemented to identify knowledge needs for station operators, engineering, work control personnel, first line supervisors, and Senior Management. These issues will then be taken to the discipline training committees and the Senior Training Advisory Committee to determine, schedule, and conduct training.

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As part of the human performance High Impact Team (HIT) efforts, a seminar for nuclear workers at Quad Cities that strengthens the site's use of questioning attitude will be developed and presented, including focus on good communications techniques.

Dynamic simulator training scenarios focusing on Reactor Water Level control are being conducted using the updated simulator model. They have been added to the current training cycle and will be completed through all shifts.

Any other actions described in the submittal represent intended or planned actions by Commonwealth Edison (ComEd) Company. They are described for the NRC's information and are not regulatory commitments.

Should you have any questions concerning this letter, please contact Mr. C.C. Peterson at (309) 654-2241, extension 3609.

Respectfully,



Joel P. Dimmette, Jr.
Site Vice President
Quad Cities Nuclear Power Station

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Quad Cities Nuclear Power Station

LICENSEE EVENT REPORT (LER)

Estimated burden per response to comply with this mandatory information collection request: 50 hrs. Reported lessons learned are incorporated into the licensing process and fed back to industry. Forward comments regarding burden estimate to the Information and Records Management Branch (1-6 133), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to the Paperwork Reduction Project (3150-0104), Office Of Management And Budget, Washington, DC 20503. If an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

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05000265

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TITLE (4)

Reactor Scram and Reactor Water Cleanup System Isolation While Returning Turbine Control Valve to Service

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MON	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
05	22	2000	2000	007	00	06	20	2000	N/A	N/A
OPERATING MODE (9)			THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR § (Check one or more) (11)							
1			20.2201(b)			20.2203(a)(2)(v)			50.73(a)(2)(i)	50.73(a)(2)(viii)
POWER LEVEL (10)			20.2203(a)(i)			20.2203(a)(3)(i)			50.73(a)(2)(ii)	50.73(a)(2)(x)
100			20.2203(a)(2)(i)			20.2203(a)(3)(ii)			50.73(a)(2)(iii)	73.71
			20.2203(a)(2)(ii)			20.2203(a)(4)		X	50.73(a)(2)(iv)	OTHER
			20.2203(a)(2)(iii)			50.36(c)(1)			50.73(a)(2)(v)	Specify in Abstract below or in NRC Form 365A
			20.2203(a)(2)(iv)			50.36(c)(2)			50.73(a)(2)(vii)	

LICENSEE CONTACT FOR THIS LER (12)

NAME

Charles Peterson, Regulatory Assurance Manager

TELEPHONE NUMBER (Include Area Code)

(309) 654-2241 ext 3609

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX

SUPPLEMENTAL REPORT EXPECTED (14)

YES		X		NO		EXPECTED SUBMISSION DATE (15)		MONTH	DAY	YEAR
(If yes, complete EXPECTED SUBMISSION DATE).										

ABSTRACT (Limit to 1400 spaces, i. e., approximately 15 single-spaced typewritten lines) (16)

On May 22, 2000, at 2159 hours, the Unit 2 reactor scrambled on a high flux signal during a return to service for an electro-hydraulic control (EHC) solenoid replacement on a turbine control valve (CV). The high flux was due to an unanticipated response of the CVs, most likely from air entrapped in the EHC system. The root cause of this event was inadequate risk assessment. Corrective actions include improving the risk assessment process to ensure available options are explored and independently reviewed, and strengthening the questioning attitude at the station.

On May 22, 2000, at 2229 hours, a reactor water cleanup system isolation occurred on low reactor water level (RWL). The actuation occurred when operators delayed the start of a reactor feed pump while awaiting an in-plant check. The root cause of the isolation was inadequate decision-making regarding restart of components. Corrective actions to prevent recurrence include updated dynamic simulator training focussing on RWL control.

The safety significance of these events was minimal. The only mitigating equipment that was not fully operable was the "D" residual heat removal service water pump. The unit was shut down using normal operating equipment. Although feedwater flow was not fully restored prior to receiving the low level isolation, the restoration was in progress and the low level signal was cleared within three minutes.

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TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

PLANT AND SYSTEM IDENTIFICATION:

General Electric - Bolling Water Reactor - 2511 MWt rated core thermal power

Energy Industry Identification System (EIS) Codes are identified in the text as [XX] and are obtained from IEEE Standard 805-1984, IEEE Recommended Practice for System Identification in Nuclear Power Plants and Related Facilities.

EVENT IDENTIFICATION:

A. PLANT CONDITIONS PRIOR TO EVENT:

Unit: 2

Event Date: May 22, 2000

Event Time: 2159 hours

Reactor Mode: 1

Mode Name: Power Operation

Power Level: 100 %

Power Operation (1) - Mode switch in the RUN position with average reactor coolant temperature at any temperature.

B. DESCRIPTION OF EVENT:

On May 21, 2000, at 2015 hours, Operations commenced performance of QCOS 5600-01, "Turbine Control Valve Fast Closure Scram Instrumentation Channel Functional Test." Step F.7 of this procedure states, "Close #1 Control Valve by depressing M.S.V.-1 TEST/C.V.-1 TEST pushbutton on 901(2)-7 Panel AND verify valve fast closes final 10%." This test is designed to cause a Turbine Control Valve (CV) [PCV] to go closed, fast-closing the last 10% from actuation of the fast-acting solenoid [SOL], and giving a 1/2 scram signal due to sensed low electro-hydraulic control (EHC) [JJ] pressure on the disc dump valve [V]. At 2020 hours, while performing this step to test the first CV, the reactor protection system (RPS) [JC] relay 590-121A did not actuate and the RPS test box light did not come on as expected. The nuclear station operator (NSO) at the control panel, who was monitoring the meter indication for the first CV while monitoring the other three CV's, initially believed he saw the #1 CV fast close. The unit supervisor (US) was also monitoring all four CVs during the evolution and did not note the #1 CV fast close. After consulting with the shift manager, it was decided to repeat the step on the first CV. The test was performed again at 2030 hours and at 2038 hours, with the RPS relay and the RPS test box responding as expected. As required by plant technical specifications, the affected RPS channel was placed in the tripped condition on Unit 2 on May 21, 2000, at 2106 hours because the #1 CV had not provided an RPS scram signal during weekly main turbine testing.

The decision was made to replace the fast-acting solenoid and the RPS switch [PS]. The risk of the activity for a unit shutdown was balanced against the risk of extending operation with a 1/2 scram inserted. The reasons for replacing both the fast-acting solenoid and the pressure switch included the fact that the NSO initially believed he saw the fast closure, although he was not absolutely certain (if the fast-acting solenoid had acted correctly, this would indicate a failure of the RPS actuating circuit such as the RPS pressure switch) and the fact that the Unit 2 pressure switches have had a history of set-point drift (these type of switches had been replaced on Unit 1).

During further discussions, it was pointed out that, although an evolution in January involving isolation of EHC to a CV was performed at 60% power, the closure and reopening of a CV is done weekly at full power through surveillance procedure QOS 5600-01. The success of the similar return to service in January in conjunction with the similarity of this activity to weekly testing of the CVs were factors that led personnel to conclude that the evolution would be successful and had an acceptable level of risk. A thorough review of the differences between the two evolutions was not conducted.

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While the procedure was being revised to allow performing the work on line, several questions were raised concerning on-line performance of the work. The previously arrived at logic and conclusions were provided by Operations personnel and were accepted by the procedure writer.

The maintenance activity was screened as high risk in accordance with WC-AA-104, "Review And Screening For High Production Risk Activities And Work Authorization." WC-AA-104 identifies that the Work Week Manager coordinates with Operations for the development of a High Production Risk Activity Mitigation Plan for such emergent work. Operations personnel developing this plan identified the scram risk as associated with removal of the incorrect fuse or another wrong component error. The RTS at power was not identified as a production risk in the plan.

On May 22, 2000, at 1330 hours, a heightened level of awareness (HLA) briefing was conducted for the work activity. The briefing was attended by the Station Manager, Operations Manager, Work Control Manager, Engineering Manager, Shift Manager, System Engineering, Operations, and maintenance personnel involved in the activity. The Station Manager, Operations Manager, and Engineering Manager raised questions concerning venting of the air in the system on the RTS. The system engineer responded that acceptable venting would be provided by holding in the CV test pushbutton for 15 seconds after the isolation valve was open. The valve has designed internal leakage that would allow air to escape when pressurized. This venting methodology differed from that used in the January 2000 RTS by in that it required additional measures to ensure venting.

At 2000 hours, the Unit 2 Supervisor conducted a second HLA briefing with the Shift Manager present and included the actions necessary to perform the RTS. The Unit 2 Supervisor covered proper communications and stressed slow operation of the EHC supply valve. The System Engineer provided direction to the NSO to hold the test button down for 15 seconds while the isolation valve was slowly opened. The solenoid replacement work was completed between 2030 hours and 2100 hours.

During the RTS, the NSO depressed the test button for the #1 CV and an in-plant operator slowly opened the EHC supply valve, hearing fluid flow as he opened the valve.

When the operator had fully opened the EHC supply valve, the other operator informed the Unit 2 NSO. The NSO held the test button in for an additional 15 seconds and then released it. The #1 Turbine Control Valve started opening as expected. No changes to EHC pressure were noted by the NSO, Unit 2 Supervisor, or by the Shift Manager when the CV was opening.

At 2159 hours, when the valve indicated approximately 50% open, the Unit 2 reactor automatically scrammed. An Average Power Range Monitor (APRM) high-high signal stemming from a reactor pressure spike caused the automatic reactor scram. The unit was placed in hot shutdown (Mode 3).

At 2200 hours, Reactor Water Level (RWL) decreased to the +8" low RWL set-point (approximately +15" indicated) due to "shrink" following the reactor scram and then increased to the +48" Reactor Feed Pump (RFP) automatic trip set-point. Following the automatic trip of the 2A and 2B RFPs, the 2A Feed Water Regulating Valve (FWRV) locked up in the closed position. The FWRV lockup was later determined to be caused by dialing down the control potentiometer demand position past the closed position. This occurred when the operators were responding to the high RWL transient following the scram. This FWRV lockup contributed to the high RWL, but had no impact on the subsequent low RWL event or recovery.

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The procedure guidance for restoring a reactor feed pump trip is contained in QCOA 3200-01, "Reactor Feed Pump Auto Trip." Specific steps are included in the procedure addressing actions associated with reactor feed pump trips from high RWL. RWL began decreasing at 2-3 inches per minute following the RFP trip and the Control Room closed the 2B RFP discharge valve in preparation for restarting the pump. At approximately 2220 hours, with RWL at 35" the NSO requested the in-plant operator to perform a pre-start check on the 2B RFP. At approximately 2224 hours, level had reached 25" when the in-plant operator reported to the Control Room that the 2B RFP seal leakage appeared excessive. As a result the Control Room did not start the 2B RFP. The Control Room began closing the 2A RFP discharge valve in preparation for starting the 2A RFP. At 2229 hours, RWL decreased to the +8" set-point for the ESF group isolation (approximately 15" indicated) that isolated the Reactor Water Cleanup (RWCU) System. The 2A RFP was then started and by 2232 hours, RWL was increasing and the +8" low RWL signal was reset.

On May 23, 2000, at 2311 hours, an Emergency Notification System (ENS) phone call was made that reported the high APRM flux and the initial +8 RWL related scrams on Unit 2. On May 23, 2000, at 0608 hours, a follow-up ENS phone call was made to report the subsequent low RWL ESF actuation.

C. CAUSE OF EVENT:

The scram resulted from an APRM high-high flux scram signal due to an unanticipated response of the CVs and EHC systems. The most likely reason for the unanticipated response was entrapped air in the EHC system following the fast-acting solenoid replacement and the incomplete purging of air from the EHC system prior to the return-to-service. This entrapped air inhibited the #1 CV from responding properly to control reactor pressure. This led to an increase in reactor pressure and reactor power and the resultant reactor scram on high reactor power. The CVs and the reactor pressure control system are very sensitive to EHC perturbations, especially when the reactor is at full power and the CVs are essentially full open. Definitive determination of the EHC and CV response is not possible because the plant does not have the installed instrumentation to support a detailed evaluation of EHC system and CV response.

The root cause identified for the APRM high-high flux scram is an inadequate risk assessment evaluating the repair and return to service of the CV fast-acting solenoid and RPS pressure switch at full power. This was due to collective overconfidence. The assessment lacked rigor and had weaknesses in documentation and implementation. Potential options were not fully explored and the risk assessment didn't fully evaluate previous similar activities. As a consequence some options were not considered or not pursued and decisions were made on incomplete or inaccurate information.

The low RWL ESF actuation occurred due to a failure to expedite restoration of the RFPs following the high APRM flux scram. Although the operators believed they were managing the time properly, the time available to start the pump could not accommodate the contingency of a deficiency found in the 2B RFP.

The root cause identified for the low RWL ESF actuation is ineffective management of RWL control due to inadequate decision-making by the operating crew.

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D. SAFETY ANALYSIS

The safety significance of the transients on May 22, 2000, was minimal. Although the spurious operation of the control valves did cause a reactor pressure spike resulting in a reactor trip, which challenged the unit's safety systems, the resulting actions are bounded by events as described in the UFSAR. The only mitigating equipment that was not fully operable was the "D" residual heat removal service water (RHRSW) pump. The other components of the residual heat removal system, including the "A," "B" and "C" RHRSW pumps, were operable. Plant safety equipment operated as designed, with the exception of the Unit 2 emergency diesel generator (EDG). The Unit 2 EDG received a spurious auto-start signal during the transfer of Unit 2 station electrical loads from on-site power to off-site power. Modifications are prepared to address this issue and are scheduled for installation on all three EDG's. The unit was shut down using normal operating equipment, and emergency core cooling systems, with the exception of the "D" RHRSW pump, were operable throughout the event. Although feedwater flow was not fully restored prior to receiving the RWL low level scram, the restoration was in progress, level never decreased below +14" indicated on the trend computer data, and the RWL low level signal was cleared within 3 minutes.

E. CORRECTIVE ACTIONS:

Corrective Actions Completed (addressing the root cause for the reactor scram):

A Station Manager memorandum describing interim actions to be performed pending revision of the appropriate work control procedures was issued June 3, 2000, with a follow-up memorandum issued June 6, 2000.

Revision 2 of the Duty Manager Responsibilities Guidelines for Quad Cities Station was issued. It provided additional guidance concerning when the Outage Control Center should be staffed. It also provided clarification of the duty team's support role for the shift manager.

Corrective Actions to be Completed (addressing the root cause for the reactor scram):

Appropriate changes to the processes and procedures associated with risk mitigation, pre-job briefs, and infrequently performed evolutions will be implemented to ensure that appropriate checks and balances are incorporated in the processes.

Documented senior manager nuclear fundamentals, with improvements based on a gap analysis, will be implemented. An action plan for improvement will be developed and implemented.

A review of recent Technical Specification events and causal factors at Quad Cities Nuclear Power Station will be developed and implemented to identify knowledge needs for station operators, engineering, work control personnel, first line supervisors, and Senior Management. These issues will then be taken to the discipline training committees and the Senior Training Advisory Committee to determine, schedule, and conduct training.

As part of the human performance High Impact Team (HIT) efforts, a seminar for nuclear workers at Quad Cities that strengthens the site's use of questioning attitude will be developed and presented, including focus on good communications techniques.

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Corrective Action to be Completed (addressing the root cause for the RWCU isolation):

Dynamic simulator training scenarios focusing on Reactor Water Level control are being conducted using the updated simulator model. They have been added to the current training cycle and will be completed through all shifts.

F. PREVIOUS OCCURRENCES:

A review of all reportable events over the last 24 months did not identify any events with root causes similar to either the root cause for the APRM scram or the root cause of the low RWL ESF actuation.

G. COMPONENT FAILURE DATA:

There were no component failures associated with this event.